TMI-2 SOLUTIONS

January 7, 2022

TMI2-RA-COR-2022-0002

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN; Document Control Desk
Washington, DC 20555-0001

Subject: License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications, Supplemental Information

Three Mile Island, Unit 2
NRC Possession Only License No. DPR-73
NRC Docket No. 50-320

References:

1. Letter TMI2-RA-COR-2021-0002, from Van Noordennen, G. P. (TMI-2 Solutions), License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications (ML21057A047)
2. Letter TMI2-RA-COR-2021-004 from Van Noordennen, G. P. (TMI-2 Solutions), Notification of "Amended Post-Shutdown Decommissioning Activities Report" (PSDAR) for Three Mile Island, Unit 2 in Accordance with 10 CFR 50.82(a)(7), Revision 4 (ML21084A229)
3. Letter TMI2-RA-COR-2021-0012 from Van Noordennen, G. P. (TMI-2 Solutions), Update 14 of Post-Defueling Monitored Storage Safety Analysis Report (ML21236A288)
4. Letter Masnik, M.T. (NRC) Proposed Possession Only License, Proposed Technical Specifications and Supporting Safety Evaluation for Post Defueling Monitored Storage at Three Mile Island Unit 2 (92003040090)
5. U.S. Nuclear Regulatory Commission (NRC). 1981. Final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste Resulting from March 28, 1979, Accident at Three Mile Island Nuclear Station, Unit 2. NUREG-0683, U.S. Nuclear Regulatory Commission, Washington, D.C. (ML19343C359)
6. U.S. Nuclear Regulatory Commission (NRC). 1989. Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste Resulting from March 28, 1979, Accident at Three Mile Island Nuclear Station, Unit 2, Final Supplement Dealing with Post-Defueling Monitored Storage and Subsequent Cleanup. NUREG-0683 Supplement No. 3, U.S. Nuclear Regulatory Commission, Washington, D.C. (ML20247F778)
7. Letter C312-91-2023 from Long R. L. (GPU Nuclear), Post-Defueling Monitored Storage Safety Analysis Report Amendment 11, (9106190266)
8. GEND-031, Submerged Demineralizer System Processing of TMI-2 Accident Waste Water, February 1983.

A001
NRR

Reference 1 requested an amendment to the Possession Only License (POL) and Appendix A, Technical Specifications (TS), of POL No. DPR-73 ("License") for Three Mile Island Nuclear Station, Unit 2 ("TMI-2"). This proposed License Amendment Request (LAR), upon approval, will revise the POL and the associated TS to support the transition of TMI-2 from a Post-Defueling Monitored Storage (PDMS) (equivalent to SAFSTOR) condition to that of a facility undergoing radiological decommissioning (DECON) pursuant to 10 CFR 50.82(a)(7).

Reference 1 Attachment 7 contained an evaluation of a high integrity container (HIC) fire. This evaluation considered the activity released from a HIC containing expended resins, assumed an unfiltered release for a 2-hour duration fire and 100 percent combustion of the contents of the HIC. This event is no longer possible as TMI-2 Solutions will use inorganic, thus non-combustible, media for processing water at TMI-2. Due to the high radiation levels from the expended media, handling and storage will be in shielded containers. Therefore Reference 1 Attachment 7 can be deleted and replaced with Attachment 1 to this letter.

Based on the elimination of this event and to determine a limiting event for TMI-2 in the DECON Phase, various TMI-2 related documents were consulted. As stated in the PSDAR Revision 4 (Reference 2). Section 3, "Description of Decommissioning Activities,"

> ... administrative, regulatory and engineering planning will occur as part of Phase 1a, while TMI-2 remains in PDMS. Upon entry into DECON, major decommissioning activities begin with Phase 1b, which entails activities necessary to complete the cleanup from the March 28, 1979, accident (i.e., source term reduction and debris material removal). Phase 1b decommissioning activities are evaluated against the potential environmental impacts analyzed in the PEIS (Programmatic Environmental Impact Statement, References 5 and 6). The objective of Phase 1b decommissioning is to achieve building and equipment decontamination to the point where general area dose rates approximate those of an undamaged reactor nearing the end of its operating life. At the completion of Phase 1b, TMI-2 will prepare for Phase 2 decommissioning which entails typical D&D activities.

Therefore, to assess a bounding event for TMI-2, several documents that form the current licensing basis for TMI-2 were reviewed.

These documents included:

1 Update 14 of Post-Defueling Monitored Storage Safety Analysis Report (Reference 3), specifically Section 8.2 Unanticipated Events Analysis

2 The NRC Technical Evaluation of Post-Defueling Monitored Storage, (Reference 4), specifically Sections 5.4.3 Accidental Atmospheric Releases and 5.4.4 Accidental Liquid Releases

3 NUREG-0683 the PEIS-Decontamination and Disposal of Radioactive Wastes Resulting from TMI-2, (Reference 5), Specifically Section 10.4 Doses from Postulated Accidents

4 NUREG-0683 Supplement 3 PEIS, Post-Defueling Storage and Cleanup (Reference 6) specifically Section 3.2.2 Offsite Dose Evaluation for Delayed Cleanup subsections 3.2.2.3 Accidental Atmospheric Releases and 3.2.2.4 Accidental Liquid Releases

Based on this review a variety of events were selected for further review based on the potential radiological significance of the event. Additionally, an offsite effluent resulting in 100 mrem Total Effective Dose Equivalent (TEDE) at the Exclusion Area Boundary was established as a limit for any event that could occur at TMI-2. Attachment 1 contains a revised TMI-2 Event Analysis and confirms the 100 mrem TEDE limit is not exceeded. Attachment 2 contains an amended No Significant Hazards Consideration (NSHC) based on the revised TMI-2 Event Analysis. This NSHC replaces the NSHC provided in Reference 1. Elsewhere in Reference 1 the description of a HIC Fire should be replaced by the events described in Attachment 1.

Attachment 3 contains a revised list of Regulatory Commitments and supersedes the list provided in Reference 1.

Because of the potential for an offsite release from TMI-2 in Phase 1B, TMI-2 Solutions in coordination with Exelon will amend the TMI Site Emergency Plan to retain the Emergency Action Levels for Radiological Effluents specific to TMI-2 and to add an EAL for a Fire as follows:

- PD-RU1 - Station release of airborne radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.
- PD-RA1 - Release of airborne radioactivity to the environment resulting in offsite dose greater than 10 mrem TEDE.
- PD-HU - Fire not extinguished within 30 minutes of detection (by fire alarm or direct observation)

A modification will be made to the initiating conditions and EALs to reflect the specific conditions at TMI-2.

After completion of Phase 1B we will confirm if there is a need to maintain these initiating conditions and EALs in Phase 2. We will notify the NRC of any changes via letter.

In addition, there are two minor changes to the proposed Technical Specifications submitted via Reference 1.

A change has been made to the No Significant Hazards Consideration to indicate that Technical Specification 3/4.4 Sealed Sources will be relocated to the Defueled Safety Analysis Report (DSAR) rather than the Decommissioning Quality Assurance Plan (DQAP).

Proposed Technical Specification 6.7.1 contained an administrative error in that the words "… as described in the…" were inadvertently deleted thus changing the meaning of this Technical Specification. This Technical Specification should read:

> 6.7.1 Written procedures shall be established, implemented, and maintained for the activities ~~necessary to maintain the PDMS condition~~ ***to be performed in Phase 1b and Phase 2*** as described in the ~~PDMS~~ SAR. Examples of these activities are:
>
> a. Technical Specification implementation.
>
> b. Radioactive waste management and shipment.
>
> c. Radiation Protection Plan Implementation.
>
> d. Fire Protection Program implementation.
>
> e. Flood Protection Program implementation.

In the event that the NRC has any questions with respect to the content of this document or wishes to obtain any additional information, please contact me at 860-462-9707.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 7, 2022.

Sincerely,

Gerard van Noordennen

Digitally signed by Gerard van Noordennen
DN: cn=Gerard van Noordennen, o=EnergySolutions, ou=Regulatory Affairs, email=gpvannoordennen@energysolutions.com, c=US
Date: 2022.01.07 15:05:39 -05'00'

Gerard van Noordennen
Senior Vice President Regulatory Affairs
TMI-2 Solutions, LLC

Attachments:

Attachment 1 – TMI-2 Event Analysis
Attachment 2 – Amended No Significant Hazards Consideration (NSHC)
Attachment 3 - List of Regulatory Commitments

cc w/Attachments
Ted Smith, NRC Project Manager
NRC Region I Administrator
NRC Lead Inspector

TMI-2 Service List
cc w/ Attachments:

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Attachment 1
TMI-2 Event Analysis

TMI-2 Potential Accidents

To determine a limiting accident for TMI-2 in the DECON Phase various TMI-2 related documents were consulted to develop a range of accidents to consider. As stated in Section 3, "Description of Decommissioning Activities," of the PSDAR Revision 4 (Reference 2) administrative, regulatory and engineering planning will occur as part of Phase 1a, while TMI-2 remains in PDMS. Upon entry into DECON, major decommissioning activities begin with Phase 1b, which entails activities necessary to complete the cleanup from the March 28, 1979 accident (i.e., source term reduction and debris material removal). Phase 1b decommissioning activities are evaluated against the potential environmental impacts analyzed in the PEIS. The objective of Phase 1b decommissioning is to achieve building and equipment decontamination to the point where general area dose rates approximate those of an undamaged reactor nearing the end of its operating life. At the completion of Phase 1b, TMI-2 will prepare for Phase 2 decommissioning which entails typical D&D activities.

Therefore, to assess a bounding event for TMI-2 several documents, (References 3 through 6), that form the licensing basis for TMI-2 were reviewed.

Based on this review a variety of events were selected for further review based on the potential radiological significance of the event. It is noted that the basis for this submittal and the commitments contained herein is the analysis for the most limiting events, (i.e., Reactor Building Fire and expended Liner Drop) for TMI-2. Descriptions and results provided for non-limiting events are intended to show the completeness of the analysis not to establish additional events or limits to consider.

Reactor Building Fire

As described in the PDMS SAR (Reference 3) the most limiting fire occurs while the Reactor Building Purge is operating. Therefore, the Reactor Building Purge is assumed to continue to operate at the nominal flowrate through the entire duration of the fire. All fire zones in the Reactor Building were analyzed with respect to resultant off-site dose. Any plateout of the airborne source term was conservatively ignored. A 99% efficiency is assumed for the HEPA filters in the Reactor Building Purge. This release to the environment resulted in a dose to the maximally exposed individual of 0.97 mrem TEDE.

However, as containment isolation will no longer be maintained during DECON phase 1B some unfiltered release is possible. Our analysis of this scenario assumes all the activity made airborne by the fire was released, without filtration, resulting in a dose to the maximally exposed individual of 97 mrem TEDE. Thus, the dose at the Exclusion Area Boundary (EAB) would be less than 100 mrem TEDE without HEPA filtration. The PDMS RB Fire Analysis conclusion will remain valid during DECON phase 1B and beyond, through administrative controls which will limit the movement of unpackaged

radioactive material in the Reactor Building to ensure no zone contains more Radioactive Material (RAM) than the limiting zone in the PDMS analysis.

The NRC has previously reviewed various Reactor Building Fire Scenarios in both NUREG-0683 Supplement 3 PEIS, Post-Defueling Storage and Cleanup Section 3.2. Delayed Cleanup (Post-Defueling Monitored Storage Followed by Completion of Cleanup) (Reference 6) and the NRC Technical Evaluation Report for PDMS (Reference 4). In particular, the NRC Technical Evaluation Report for PDMS (Reference 4) reviewed the TMI-2 PDMS Reactor Building Fire Analysis, with the potential failure of the filtered path, and concluded a release of 0.65 curie of cesium-137 and 0.03 curie of strontium-90 would occur resulting in an offsite dose consequence of 49 mrem whole body 51 mrem bone, also less than 100 mrem TEDE at the EAB.

Fire Outside of the Reactor Building

PEIS Section 10.4.1.2 (Reference 5) evaluated a fire occurring in combustible radioactive material. As some combustible radioactive material will be staged prior to packaging in non-combustible containers, similar methodology as described in the PDMS SAR for a Reactor Building fire was performed for areas outside the Reactor Building. This analysis is a supplement for areas outside the Reactor Building such as the AFHB complex, Chemical Cleaning Building (CCB), Turbine Building or other as yet unidentified areas which may contain radioactive material storage and handling areas or have removable contamination source terms that could be resuspended in a fire due to decommissioning activities. The following scenarios were evaluated:

- An area fire releasing removable contamination in the AFHB, CCB, Turbine Building or other areas/facilities with or without HEPA filtration as a ground release.

- Combustible materials fire releasing contamination from materials staged in areas outside the Reactor Building such as the AFHB, CCB, Turbine Building or Yard Area.

The source term radionuclide mixes described in the Fire Protection Program Evaluation (FPPE) were taken from the Reactor Building Fire Analysis and are used in this calculation. The Fire Zone source terms in the Reactor Building, Auxiliary and Fuel Handling Buildings were decay corrected to January 1, 2021 and source term limits for areas outside the Reactor Building were developed from a normalized composite mix to ensure the 10 CFR 20 limits to a member of the public at or beyond the site boundary (i.e., exclusion area boundary (EAB)), would not be exceeded in the event of a fire. The AFHB has various doors and openings, such that in the event of a fire 100% of the activity would not be released through the plant vent and would not be an elevated release. Buildings other than the Reactor Building are also not elevated releases. Therefore, a ground release emergency X/Q is calculated for these areas since the

Attachment 1
TMI-2 Event Analysis

Reactor Building Fire Analysis X/Q is for the plant stack, which is an elevated release.

Source term limits for area removable surface contamination are provided in Table 1.

Table 1 - Calculated 1 mrem EAB Area Activity Limits for Areas Outside Rx Bld

Area Contamination Limits		
Nuclide	***HEPA Filter Ground Release Ci***	***No HEPA Filtered Ground Release Ci***
Sr-90	5.62E+02	5.62E+00
Cs-137	1.63E+03	1.63E+01
Pu-238	4.35E-02	4.35E-04
Pu-239	3.05E-01	3.05E-03
Pu-240	1.48E-01	1.48E-03
Pu-241	1.56E+00	1.56E-02
Am-241	5.56E-01	5.56E-03
Totals	2.20E+03	2.20E+01

The HEPA filter source term limits for an elevated (Rx/AFHB) release are provided in Table2.

Table 2 - HEPA Filter Elevated Release 1 mrem and 80 mrem EAB Source Term Limit

Composite Release	***EAB Air Conc pCi/m3***	***Release Rate Q pCi/sec***	***Total 1 mrem Ci on Filter***	***Total 80 mrem Ci on Filter***
Sr-90	3.56E+02	4.64E+05	1.7E-03	1.3E-01
Cs-137	1.03E+03	1.35E+06	4.9E-03	3.9E-01
Pu-238	2.75E-02	3.59E+01	1.3E-07	1.0E-05
Pu-239	1.93E-01	2.52E+02	9.1E-07	7.3E-05
Pu-240	9.34E-02	1.22E+02	4.4E-07	3.5E-05
Pu-241	9.90E-01	1.29E+03	4.6E-06	3.7E-04
Am-241	3.52E-01	4.59E+02	1.7E-06	1.3E-04
Totals			6.5E-03	5.2E-01

The corresponding dose rate cut off limits on the HEPA filter housing are provided in Table 3.

Table 3 - Elevated HEPA Filter Dose Rate Limits

Cut Off	***Location***	***Contact mR/hr***	***30 cm mR/hr***	***6 feet mR/hr***
1 mrem	*Side*	6.22E+00	2.05E+00	2.29E-01
80 mrem	*Side*	4.95E+02	1.63E+02	1.82E+01

The source term limits for activity on a ground release HEPA filter are provided in Table 4.

Table 4 – 1 mrem and 80 mrem EAB Source Term Limit for a ground release HEPA Filter Failure

Composite Release	EAB Air Conc pCi/m3	Release Rate Q pCi/sec	Total 1 mrem Ci on Filter	Total 80 mrem Ci on Filter
Sr-90	3.56E+02	2.34E+05	8.4E-04	6.7E-02
Cs-137	1.03E+03	6.80E+05	2.4E-03	2.0E-01
Pu-238	2.75E-02	1.81E+01	6.5E-08	5.2E-06
Pu-239	1.93E-01	1.27E+02	4.6E-07	3.7E-05
Pu-240	9.34E-02	6.15E+01	2.2E-07	1.8E-05
Pu-241	9.90E-01	6.51E+02	2.3E-06	1.9E-04
Am-241	3.52E-01	2.32E+02	8.3E-07	6.7E-05
Totals			3.3E-03	2.6E-01

The corresponding dose rate cut off limits on the HEPA filter housing are provided in Table 5.

Table 5 – Non-Elevated HEPA Filter 1 mrem and 80 mrem Cut-Off Dose Rates

Cut Off	Location	Contact mR/hr	30 cm mR/hr	6 feet mR/hr
1 mrem	Side	3.05E+00	1.00E+00	1.12E-01
80 mrem	Side	2.54E+02	8.36E+01	9.35E+00

The source term limits for uncompacted combustible DAW stockpiled outside containers are provided in Table 6.

Table 6 - 1 mrem EAB TEDE Uncontained Combustible Waste Activity Limits for HEPA Filtered and Non-HEPA Filtered Ground Releases

Nuclide	EAB Concentration, pCi/m^3	Total Ci Limit on DAW Ground HEPA Filtered	Total Ci Limit on DAW Ground Not HEPA Filtered
Sr-90	3.56E+02	5.62E+02	5.62E+00
Cs-137	1.03E+03	1.63E+03	1.63E+01
Pu-238	2.75E-02	4.35E-02	4.35E-04
Pu-239	1.93E-01	3.05E-01	3.05E-03
Pu-240	9.34E-02	1.48E-01	1.48E-03
Pu-241	9.90E-01	1.56E+00	1.56E-02
Am-241	3.52E-01	5.56E-01	5.56E-03
Totals		2.20E+03	2.20E+01

The corresponding dose rate cut-off values are provided in Table 7.

Table 7 - Dose Rate Cut Off Limits for Uncompacted DAW of Various Sizes Stored Outside Containers Ground Release

	Uncontained DAW Ground Release No HEPA		
Pile Size	***contact mR/hr***	***30 cm mR/hr***	***1 meter mR/hr***
6x4x4 feet (B-25 size)	7.72E+03	3.98E+03	1.35E+03
20x8x7 (Intermodal Size)	7.83E+02	5.81E+02	3.25E+02
19.17x7.08x5.08 feet (Intermodal Size)	1.16E+03	7.65E+02	3.94E+02
18.96x7.33x6.75 (Sealand Size)	7.38E+02	5.77E+02	3.45E+02

The source term limits for uncompacted DAW stored in open containers are provided in Table 8.

Table 8 - Uncompacted DAW Container Cut Off Values for Ground Release Not HEPA Filtered Storage While Open

	Dose Rates for Contained DAW Ground Release No HEPA			
Container	***contact mR/hr***	***30 cm mR/hr***	***1 meter mR/hr***	***3 meter mR/hr***
B-25	1.71E+03	9.82E+02	2.83E+02	3.33E+02
Roll Off	2.47E+02	1.76E+02	9.22E+01	2.81E+01
Yard Intermodal 20ft	2.10E+02	1.49E+02	7.83E+01	2.38E+01
Sealand Cargo Container	1.38E+02	1.15E+02	6.81E+01	2.20E+01

The corresponding dose rate cut off values on the containers are provided in The Table 9 to ensure that a fire and unfiltered release will not result in greater than 1 mrem TEDE at the EAB.

Table 9- Dose Rate Cut Off Limits for Uncompacted DAW Stored in Open Containers Ground Release

	Contained DAW Ground Release No HEPA			
Container	***contact mR/hr***	***30 cm mR/hr***	***1 meter mR/hr***	***3 meter mR/hr***
B-25	1.71E+03	9.82E+02	2.83E+02	3.33E+02
Roll Off	2.47E+02	1.76E+02	9.22E+01	2.81E+01
Yard Intermodal 20 ft	2.10E+02	1.49E+02	7.83E+01	2.38E+01
Sealand Size	1.38E+02	1.15E+02	6.81E+01	2.20E+01

Because localized regions of contamination (i.e., Hot Spots) can bias the contact dose rate reading, average contact values or more distant dose rate readings (e.g., 1 meter or 2 meter) should be used to gauge the source terms on HEPA filter housings, DAW stockpiles, and Containers. As long as the radionuclide mix approximates the mix used to calculate the cut off dose rates, the activity is proportional to the dose rate.

The dose rate cut-off limits will be used periodically to monitor HEPA filter dose rates to determine when they should be changed out in order to ensure that in the event of a fire the EAB TEDE limit will not be exceeded.

The Table 7 dose rate cut-off values for combustible uncompacted Dry Active Waste are limits for DAW stored outside fire resistant containers in areas will not result in greater than 1 mrem TEDE at the EAB in the event of a fire and unfiltered release.

The Table 9 dose rate cut-off values for combustible uncompacted Dry Active Waste are limits for DAW inside open fire-resistant containers to ensure that a fire and unfiltered release will not result in greater than 1 mrem TEDE at the EAB.

Events Involving Ion-Exchange Resins

Fire

A water processing system will be installed during DECON to process water used during the DECON phase of TMI-2. In TMI-2 Solutions letter TMI2-RA-COR-2021-0002 "License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications" dated February 19, 2021 (Reference 1) a HIC containing expended organic resins is assumed to be involved in an outside fire. This fire is unrealistic for several reasons.

- Inorganic resins, instead of organic resins, will be used to capture the radioisotopes in the high activity water.
- Resin liners will not be handled in the open air; instead expended resin liners will be transferred to a transport container inside the Chemical Cleaning Building (CCB) for transport to an onsite storage location in a different building where they will be transferred into a secure shielded container.
- The proposed water processing system will be placed in the CCB which contains a HEPA Filtered ventilation system.

The event described in reference 1 is no longer considered credible.

Liner Drop

NUREG 0683 (Reference 5) Section 10.4.1.3 Breach of a Waste-Containing Package considered the dropping of a waste package and the resulting breach and release of a portion of the contents of the package, a credible accident. During Phase 1b the

dropping of a High Activity water processing system liner was analyzed for the current mix of radionuclides at TMI-2 and liner loading.

This analysis uses the Unit 2 submerged spent demineralizer zeolite canister source terms reported in GEND-31 Table 7 (Reference 8) decay corrected to January 1, 2021 to calculate the potential airborne source terms from a drop of a zeolite HIC at 80% of the Class C limit. These source terms were then adjusted to a 100 mrem TEDE at or beyond the Exclusion Area Boundary (EAB). The release activity from a drop of an 8-120 HIC filled with zeolite was modeled as a puff release. The activity released in a drop is calculated by multiplying the source terms by a conservative airborne release fraction (ARF) of 1E-04 from NUREG 0683 Vol 1 (Reference 5), Section 8.1.4.2 Package-Handling Accidents for dewatered zeolites, and resins, accident sludges, and filter cartridges. The pCi/sec release rate is calculated by converting the source term to pCi, then dividing the source term by 3600 sec/hr. The airborne concentration at the EAB is calculated by multiplying the release rate in pCi/sec by the X/Q. The ground concentration is calculated by multiplying the release rate by the D/Q.

A ground release X/Q of 1.52E-03 sec/m3 was used for the Chemical Cleaning Building (CCB), the D/Q was 1.52E-05 sec/m2.

Assuming the zeolite is sluiced to an 8-120 HIC results in approximately 100 cubic feet of zeolite in the liner. As these liners will need to be transferred from the CCB for temporary storage and/or disposal a calculation was performed to determine what zeolite HIC activity limits would result in an offsite dose of 100 mrem TEDE at the EAB if a liner was dropped in a location that did not have HEPA filtered ventilation. The results of this calculation are provided below and represent the maximum permissible activity on a zeolite HIC. A HIC would have to be well above the Class C limit to have a drop accident result in a release contributing 100 mrem TEDE at the EAB.

Nuclide	No HEPA Limit Ci
Sr-90	1.00E+03
Cs-134	1.20E-03
Cs-137	1.82E+04
Pu-238	8.99E-10
Pu-239	8.54E-09
Pu-240	4.12E-09
Pu-241	6.92E-09
Am-241	1.59E-08
Total	**1.92E+04**

Attachment 1
TMI-2 Event Analysis

Release of Resins from Make-up and Purification System Demineralizers

TMI-2 Letter (Reference 7) Post-Defueling Monitored Storage Safety Analysis Report, Amendment 11, Letter from R. L. Long, GPU Nuclear Corporation, to NRC. dated April 19, 1991, GPU Document C312-91-2023) calculated the dose resulting from the release of resins from a makeup and purification demineralizer vessel rupturing non-mechanistically and the contents spilling onto the floor of the cubicle. The inventory of the demineralizer included 530 curies of cesium- 137, 100 curies of strontium-90/yttrium-90, and 1.1 pounds (0.5 kilograms) of fuel. TMI-2 used an airborne release factor of 0. 0001 which resulted in an estimated 0.45-millirem dose to the maximally exposed individual from a HEPA filtered release.

The NRC staff also analyzed the release of contaminated resins from a ruptured demineralizer unit in the Auxiliary Building in Reference 4. The inventory of the demineralizer included approximately 100 curies of strontium-90, 530 curies of cesium-137, and 500 grams of fuel. The primary isotopes in the fuel are plutonium-239, plutonium-240, plutonium-241, samarium-151, europium-154, and europium-155. An airborne release fraction of 0 .0005 was used. In addition, it was assumed that all airborne activity would be filtered by the HEPA filters in the AFHB ventilation system before it was released into the atmosphere. The calculated doses to the maximally exposed individual were 0. 20 millirem for the total body and 0. 25 millirem to the critical organ (the bone) from inhalation and external exposure which is very similar to result from the TMI-2 letter referenced above when you consider the difference in the airborne release factor.

Thus, in the event of an unfiltered release, the release is bounded by a Reactor Building Fire or a Liner Drop.

During Phase 1b any work involving removal of the makeup and purification demineralizer resins will be performed using good ALARA practices in accordance with the Radiation Protection Plan.

HEPA Filter Failure

Reference 5 and 6 considered the offsite dose consequence of a high-efficiency particulate air (HEPA) filter failure. The probability of such a failure is low, but a bounding case has been addressed for TMI2. Analysis has been completed, and periodic monitoring of HEPA filters for areas with significant radioactive material (including the Reactor, Auxiliary and Fuel Handling Buildings) will be performed. Based on the analysis, limits will be established to ensure filters are replaced before a failure could result in an event where the offsite dose could exceed the 100 mr TEDE limit.

Attachment 1
TMI-2 Event Analysis

Accidental Liquid Spill

Failure of a Processed Water Storage Tank

Two outside processed water storage tanks with a capacity of 500,000 gallons each have been constructed. A criterion for storage of processed water in these tanks is that the content of radioactivity stored in each outside processed water storage tank should be limited such that a tank failure would not result in greater than 10 CFR Part 20 Table II, Col. 2 concentrations at the nearest drinking water intake for combined radionuclides as a function of actual tank volume. In order to assure conformance with this criterion, several assumptions and calculations are required. Details of the assumptions used by the NRC staff for this storage criterion are discussed in Section 7.2.4.2 of Reference 5. In equation form this criterion requires that the storage of radioactivity in an outside storage tank be limited such that in the event of a rupture:

River Concentration Limit at Drinking Water Intake

$$\sum \frac{C_i}{MPC_i} \leq 1$$

where:

Ci = concentration (μCi/mL) of the ith nuclide at the nearest downstream drinking water intake (Brunner Island)

MPCi = maximum permissible concentration currently termed the Effluent Concentration (EC) (10 CFR Part 20, Table II, Col. 2 – μCi/mL)

The river concentration at the nearest drinking water intake (Ci, – μCi/mL) is determined by:

River Concentration at Nearest Drinking Water Intake

$$C_i = \frac{0.5\ A_i}{Q \Delta t\ 28316.859}$$

where:

C_i = River Concentration at nearest drinking water intake

Q = minimum river flow rate to overtop Red Hill Dam (16,000 ft^3/sec)

Δt = release period (2 hours)

Ai = tank activity prior to rupture (microCuries)

0.5 = fraction of the tank volume that discharges to the river

28316.85 = ml per cubic foot

Note that River Concentration Limit at Drinking Water Intake is dependent upon the activity of each radionuclide in the mix and must be recalculated for each radionuclide mix in the tank because each nuclide has a different effluent Concentration Limit.

The EC equivalent tank activity for each nuclide can be calculated by using the below equation where Ci equals the 10 CFR 20 Effluent Concentration for the nuclide and solving for Ai.

Tank Activity (µCi) Equal to 1 EC at Nearest Drinking Water Intake

$$\frac{C_i x\ Q\Delta t\ 28316.8466}{0.5} = A_i$$

where:

C_i = nuclide 10 CFR 20 Effluent Concentration

Q = minimum river flow rate to overtop Red Hill Dam 16,000 ft^3/sec)

Δt = release period (2 hours, 7200 sec)

Ai = tank activity prior to rupture (microCuries)

0.5 = fraction of the tank volume that discharges to the river

28316.85 = ml per cubic foot

The tank activities equal to 1 EC at the nearest drinking water intake calculated using the above equation are provided below.

1 Effluent Concentration at Nearest Drinking Water PWST Activity Limits

Nuclide	C_i EC µCi/ml	Tank Limit Ci	Full Tank µCi/ml
H-3	1.00E-03	6.52E+03	3.45E+00
C-14	3.00E-05	1.96E+02	1.03E-01
Mn-54	3.00E-05	1.96E+02	1.03E-01
Fe-55	1.00E-04	6.52E+02	3.45E-01
Ni-63	1.00E-04	6.52E+02	3.45E-01
Co-60	3.00E-06	1.96E+01	1.03E-02
Sr-90	5.00E-07	3.26E+00	1.72E-03
Tc-99	1.00E-05	6.52E+01	3.45E-02
Ru-106	3.00E-06	1.96E+01	1.03E-02
Sb-125	3.00E-05	1.96E+02	1.03E-01
I-129	0.00E+00	0.00E+00	0.00E+00
Cs-134	9.00E-06	5.87E+01	3.10E-02

Attachment 1
TMI-2 Event Analysis

Nuclide	C_i EC µCi/ml	Tank Limit Ci	Full Tank µCi/ml
Cs-137	1.00E-06	6.52E+00	3.45E-03
Ce-144	3.00E-06	1.96E+01	1.03E-02
Pu-238	2.00E-08	1.30E-01	6.89E-05
Pu-241	1.00E-06	6.52E+00	3.45E-03
Am-241	2.00E-08	1.30E-01	6.89E-05

Thus, Tank Activity will be limited to S Ai < 1

Attachment 2
Amended No Significant Hazards Consideration (NSHC)

3.3 No Significant Hazards Consideration (NSHC)

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," TMI-2 Solutions LLC, proposes an amendment to the Possession Only License (POL) and Technical Specifications, of POL No. DPR-73 for Three Mile Island Nuclear Station, Unit 2 ("TMI-2").

This proposed LAR, upon approval, will revise the POL and the associated TS to support the transition of TMI-2 from PDMS to that of a facility undergoing decommissioning. The proposed amendment would revise the POL and TS to support Phase 1b and Phase 2 activities associated with achieving the removal of all Debris Material, its transfer to dry cask storage at an Independent Spent Fuel Storage Installation (ISFSI), or to a suitable waste storage area, and the relocation of various requirements to the TMI-2 DQAP and Sealed Sources to the Defueled Safety Analysis Report (DSAR).

As noted in a letter from the NRC to GPU Nuclear dated February 13, 2013, (Reference 1) the equivalent to the certificate of cessation of operations was determined to be the NRC's issuance of TMI-2 License Amendment 45, converting the TMI-2 operating license to a possession only license. This amendment was granted on September 14, 1993 (Reference 30) and establishes that date as the date that TMI-2 is considered to have submitted certification of permanent cessation of operations.

The proposed changes to the POL and TS, for deletion or revision, are in accordance with 10 CFR 50.36(c)(1) through 10 CFR 50.36(c)(5). The proposed changes also include a renumbering of pages and sections, where appropriate, to condense and reduce the number of pages in the TS without affecting the technical content.

The existing TMI-2 TS contain Limiting Conditions for PDMS that provides the functional capability of equipment required for safe operation of the facility. The current TS are only applicable with TMI-2 in the PDMS condition. Limiting Conditions for PDMS and associated Surveillance Requirements (SRs) that will not apply in Phase 1b or Phase 2 are being proposed for deletion. The remaining portions of the TS are being proposed for revision and will continue to provide an acceptable level of control for the TMI-2 facility as it undergoes decommissioning.

TMI-2 Solutions has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes would revise the TMI-2 POL and TS by deleting or modifying certain portions of the TS that are no longer applicable to TMI-2 as it transitions from PDMS to decommissioning. This change is consistent with the criteria set forth in 10 CFR 50.36 for the contents of TS.

The Phase 1a condition is a continuation of the PDMS condition. No major decommissioning activities will occur in Phase 1a. As discussed in Section 2 "Detailed Description and Basis for the Changes" of this proposed amendment, the radiological consequences associated with the fire inside containment unanticipated event does not exceed the applicable limits of 10 CFR 100.11 and the EPA PAGs.

Following Phase 1a, TMI-2 will enter Phase 1b and Phase 2. During Phase 1b and Phase 2, major decommissioning activities as defined in 10 CFR 50.2 will be performed. As discussed in Attachment 1 to this letter, none of the events evaluated exceed the applicable limits of 10 CFR 100.11 and the EPA PAGs.

During Phase 1a, containment isolation is maintained to assure the containment is properly maintained as a contamination barrier for the residual contamination which resides inside the containment.

There are no postulated accidents that can occur inside of the Reactor Building (RB) during Phase 1b or Phase 2 that result in the dose at the site boundary exceeding the limits of 10 CFR 100.11 and the EPA PAGs including such times as when the containment engineered access equipment hatch is open. The D&D process includes many evolutions that will require the equipment hatch and other RB access points to be open to allow movement of equipment, waste, and other materials into and out of the RB. The Radiation Protection Plan (RPP) will identify the controls that will be implemented through procedures during D&D activities occurring inside of the RB. Implementation of these procedures take into account detailed work planning, and execution of the D&D work and support activities, including measures to maintain occupational dose As Low As Reasonably Achievable (ALARA) and below the occupational dose limits in 10 CFR Part 20 during decommissioning.

Procedures associated with Phase 1b will be developed to retrieve the remaining core debris and decontaminate high radiation areas. Phase 2 procedures will also be developed; however, the focus of these procedures is related to performing D&D operations in a facility which has not experienced an accident.

The deletion of TS 3/4.1 does not cause a change in facility conditions, design function, or analysis that verifies the ability of SSCs to perform a design function. The function of the containment is to contain residual contamination inside the containment during decommissioning remains unchanged. During Phase 1b the RPP and associated implementing procedures will provide the controls necessary to manage residual contamination. As such the containment continues to function as a

contamination barrier. Airborne radiation monitoring will be provided at the engineered containment openings (e.g., Equipment Hatch Opening). With the construction of the engineered openings in containment the RB breather no longer provides a preferred path to the atmosphere. No credit is taken for the containment as a pressure containing boundary and therefore unfiltered leak rate testing of the containment is no longer applicable.

The dose at the site boundary associated with any of the events evaluated in Attachment 1 does not exceed the requirements of 10 CFR 100.11, and the EPA PAGs.

Therefore, the deletion of TS 3/4.1 "Containment" does not involve a significant increase in the probability or consequences of an accident previously evaluated.

TS 3/4.2 "Reactor Vessel Fuel" establishes a Safe Fuel Mass Limit (SFML) for the PDMS condition, which ensures that the amount of core debris that may be removed from the Reactor Vessel (RV) or rearranged in the RV during PDMS does not exceed 42kg. This SFML is specified to ensure subcriticality even after dual errors.

The deletion of TS 3/4.2 does not cause a change in facility conditions, design function, or analysis that verifies the ability of SSCs to perform a design function. A calculation is presented as Attachment 5 (of Reference 1), which provides the basis to increase the SFML from 42 kg to 1200 kg. The result of this calculation demonstrates that the entire mass of the core debris material cannot be configured into an arrangement whereby a criticality event is possible and that K_{eff} does not exceed 0.95.

Therefore, the deletion of TS 3/4.2 "Reactor Vessel Fuel" does not involve a significant increase in the probability or consequences of an accident previously evaluated.

In Phase 1a, TS 3/4.3 "Crane Operations" prohibits loads in excess of 50,000 lbs. from travel over the RV.

The deletion of TS 3/4.3 does not cause a change in facility conditions, design function, or analysis that verifies the ability of SSCs to perform a design function. As discussed in Section 2 "Detailed Description and Basis for The Changes," for Phase 1b and Phase 2, TMI-2 Solutions will develop a hoisting and rigging program that addresses movement of loads at TMI-2. The purpose of the hoisting and rigging program is to define the minimum requirements for the safe operations of cranes and hoists. The hoisting and rigging program will provide as applicable, detailed requirements for training and qualification of personnel, inspection and maintenance of cranes or hoists, the safe use of rigging equipment as well as direction for performing Non-Standard Lifts in order to ensure that lifting operations are performed in a safe manner. A lift plan will be developed for all lifts as directed by the hoisting and rigging program where a load drop or load impingement could contribute to release or dispersal of radioactive material to the environment which could exceed the threshold for an unusual event.

Attachment 2
Amended No Significant Hazards Consideration (NSHC)

Implementation of the hoisting and rigging program provides a defense in depth approach to preventing a load drop from occurring. Crane design features such as load cells, and travel stops, will be employed as required to ensure safe travel paths. Barriers will be provided as per the lift plan, as required to precludethe effects of a load drop.

A calculation has been performed (Attachment 5 of Reference 1) that assesses increasing the Safe Fuel Mass Limit (SFML) from 42 kg to approximately 1200 kg. The analysis states that it is not credible to have 1200 kg U in an idealized configuration for criticality to occur. There are no credible operational upsets to realize the ideal configuration but even in the event that the upset occurs, it would require fissile mass in excess of that analyzed, which is greater than what is anticipated.

Based on the above the PDMS Quality Assurance Program for TMI-2 will also be modified from:

> Lifting and Handling activities, including testing and surveillance of cranes and rigging components where the equipment and activities involve the handling or movement of radioactive material where a load drop or load impingement could contribute to unplanned release or dispersal of radioactive material or where such activity involves the movement of loads over the Reactor Vessel, or the handling of material that could contain Special Nuclear Material.

To:

> Lifting and Handling activities for all lifts where a load drop or load impingement could contribute to release or dispersal of radioactive material to the environment which could exceed the threshold for an unusual event.

Therefore, the deletion of TS 3/4.3 "Crane Operations" does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The TMI-2 sealed sources are maintained at TMI-1 and managed by Exelon under a program compliant with the requirements of 10 CFR 70.39(c). Deleting TS 3/4.4 "Sealed Sources" from the TMI-2 TS and relocating the TS requirements to the DSAR does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The deletion of TS definitions and rules of usage and application that will not be applicable during Phase 1b and Phase 2 decommissioning, has no impact on facility structures, systems, and components (SSCs) or the methods of operation of such SSCs.

The proposed relocation of certain administrative requirements as allowed by Administrative Letter 95-06 (Reference 6) do not affect operating procedures or administrative controls that have the function of ensuring the safe management of

Debris Material or decommissioning of the facility.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes to delete and/or modify the TS does not create the possibility of a new or different kind of accident from that previously evaluated. The removal of the TS applicable in Phase 1a cannot result in different or more adverse accidents than previously evaluated because there are no new credible failure mechanisms, or accident initiators not considered in the design and licensing basis for Phase 1b.

Following Phase 1a, TMI-2 will enter Phase 1b and Phase 2. During Phase 1b and Phase 2, major decommissioning activities as defined in 10 CFR 50.2 will be performed. As discussed in Attachment 1 to this letter these events have been described in various License Basis Documents thus a new or different kind of accident from any accident previously evaluated has not been created.

During Phase 1a, containment isolation is maintained to assure the containment is properly maintained as a contamination barrier for the residual contamination which resides inside the containment.

There are no postulated accidents that can occur inside of the RB during Phase 1b or Phase 2 that result in the dose at the site boundary exceeding the limits of 10 CFR 100.11 and the EPA PAGs including such times as when the containment engineered access equipment hatch is open. The D&D process includes many evolutions that will require the equipment hatch and other RB access points to be open to allow movement of equipment, waste, and other materials into and out of the RB. The RPP will identify the controls that will be implemented through procedures during D&D activities occurring inside of the RB. Implementation of these procedures take into account detailed work planning, and execution of the D&D work and support activities, including measures to maintain occupational dose As Low As Reasonably Achievable (ALARA) and below the occupational dose limits in 10 CFR Part 20 during decommissioning. Procedures associated with Phase 1b will be developed to retrieve the remaining core debris and decontaminate high radiation areas. Phase 2 procedures will also be developed; however, the focus of these procedures is related to performing D&D operations in a facility which has not experienced an accident.

The deletion of TS 3/4.1 "Containment" does not cause a change in facility conditions, nor does it cause a change in design function. The function of the containment is to maintain residual contamination during Phase 1a remains unchanged. During Phase 1b and 2, the RPP and associated implementing procedures will provide the controls

necessary to manage residual contamination. As such, the containment continues to function as a contamination barrier. Airborne radiation monitoring will be provided at the engineered containment openings (e.g., Equipment Hatch Opening). Procedures are utilized to control routine containment access. With the construction of the engineered openings in containment the RB breather no longer provides a preferred path to the atmosphere. No credit is taken for the containment as a pressure containing boundary and therefore unfiltered leak rate testing of the containment is no longer applicable.

The dose at the site boundary associated with the events described in Attachment 1 to this letter does not exceed the requirements of 10 CFR 100.11, as well as the EPA PAGs.

Therefore, the deletion of TS 3/4.1 "Containment" does not create the possibility of a new or different kind of accident from any accident previously evaluated relative to Phase 1b or Phase 2.

TS 3/4.2 "Reactor Vessel Fuel" establishes a Safe Fuel Mass Limit (SFML) for the PDMS condition, which ensures that the amount of core debris that may be removed from the RV or rearranged in the RV during PDMS does not exceed 42kg. This SFML limit is specified to ensure subcriticality even after dual errors.

The deletion of the TS does not cause a change in facility conditions, nor does it cause a change in design function. A calculation is presented as Attachment 5 (of Reference 1), which provides the basis to increase the SFML from 42 kg to 1200 kg. The result of this calculation demonstrates that the entire mass of the core debris material cannot be configured into an arrangement whereby a criticality event is possible and that K_{eff} does not exceed 0.95.

Therefore, the deletion of TS 3/4.2 "Reactor Vessel Fuel" does not create the possibility of a new or different kind of accident from any accident previously evaluated relative to Phase 1b or Phase 2.

As part of the PDMS condition, loads in excess of 50,000 lbs. are prohibited from travel over the RV. The deletion of TS 3/4.3 "Crane Operations" does not cause a change in facility conditions nor does it cause a change in design function.

As discussed in Section 2 "Detailed Description and Basis for The Changes," (of Reference 1) for Phase 1b and Phase 2, TMI-2 Solutions will develop a hoisting and rigging program that addresses movement of loads at TMI-2. The purpose of the hoisting and rigging program is to define the minimum requirements for the safe operations of cranes and hoists. The hoisting and rigging program will provide detailed requirements as applicable for training and qualification of personnel, inspection and maintenance of cranes or hoists, the safe use of rigging equipment as well as direction for performing Non-Standard Lifts in order to ensure that lifting operations are performed in a safe manner. A lift plan will be developed for all lifts as directed by the

hoisting and rigging program where a load drop or load impingement could contribute to release or dispersal of radioactive material to the environment which could exceed threshold for an unusual event.

Implementation of the hoisting and rigging program provides a defense in depth approach to preventing a load drop from occurring. Crane design features such as load cells, and travel stops, will be employed as required to ensure safe travel paths. Barriers will be provided as required to preclude the effects of a load drop.

A calculation has been performed (Attachment 5 of Reference 1) that assesses increasing the Safe Fuel Mass Limit (SFML) from 42 kg to approximately 1200 kg. The analysis states that it is not credible to have 1200 kg U in an idealized configuration for criticality to occur. There are no credible operational upsets to realize the ideal configuration but even in the event that the upset occurs, it would require fissile mass in excess of that analyzed, which is greater than what is anticipated, in addition to a greatly reduced impurity concentration to present a criticality hazard.

Therefore, the deletion of TS 3/4.3 "Crane Operations" does not create the possibility of a new or different kind of accident from any accident previously evaluated relative to Phase 1b or Phase 2.

The TMI-2 sealed sources are maintained at TMI-1 and managed by Exelon under a program compliant with the requirements of 10 CFR 70.39(c). Deleting TS 3/4.4 "Sealed Sources" from the TMI-2 TS and relocating the TS requirements to the DSAR does not create the possibility of a new or different kind of accident from any accident previously evaluated relative to Phase 1b or Phase 2.

The proposed change will not create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the licensing bases documents. Decommissioning operations in Phase 1b and Phase 2 are bounded by the events described in Attachment 1.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated relative to Phase 1b or Phase 2.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes would revise the TMI-2 POL and TS by deleting or modifying certain portions of the TS that are no longer applicable to TMI-2 as it transitions from PDMS to decommissioning. This change is consistent with the criteria set forth in 10 CFR 50.36 for the contents of TS.

The Phase 1a condition is a continuation of the PDMS condition. No major decommissioning activities will occur in Phase 1a. As discussed in Section 2 "Detailed Description and Basis for the Changes" (of Reference 1) of this proposed amendment, the radiological consequences associated with the fire inside containment, unanticipated event, does not exceed the applicable limits of 10 CFR 100.11 and the EPA PAGs.

Following Phase 1a, TMI-2 will enter Phase 1b and Phase 2. During Phase 1b, major decommissioning activities as defined in 10 CFR 50.2 will be performed. Based on the consequences of the postulated events in Attachment 1 of this letter, none of the events evaluated involve a significant reduction in a margin of safety.

During Phase 1a, containment isolation is maintained to assure the containment is properly maintained as a contamination barrier for the residual contamination which resides inside the containment.

There are no postulated accidents that can occur inside of the RB during Phase 1b or Phase 2 that result in the dose at the site boundary exceeding the limits of 10 CFR 100.11 and the EPA PAGs including such times as when the containment engineered access equipment hatch is open. The D&D process includes many evolutions that will require the equipment hatch and other RB access points to be open to allow movement of equipment, waste, and other materials into and out of the RB. The RPP will identify the controls that will be implemented through procedures during D&D activities occurring inside of the RB. Implementation of these procedures take into account detailed work planning, and execution of the D&D work and support activities, including measures to maintain occupational dose As Low As Reasonably Achievable (ALARA) and below the occupational dose limits in 10 CFR Part 20 during decommissioning. Procedures associated with Phase 1b will be developed to retrieve the remaining core debris and decontaminate high radiation areas. Phase 2 procedures will also be developed; however, the focus of these procedures is related to performing D&D operations in a facility which has not experienced an accident.

The deletion of TS 3/4.1 "Containment" does not exceed or alter a design basis or safety limit. The function of the containment which is to maintain residual contamination during Phase 1a and 2 remains unchanged. During Phase 1b the RPP and associated implementing procedures will provide the controls necessary to manage residual contamination. As such the containment continues to function as a contamination barrier. Airborne radiation monitoring will be provided at the engineered containment openings (e.g., Equipment Hatch Opening). Procedures are utilized to control routine containment access. With the construction of the engineered openings in containment the RB breather no longer provides a preferred path to the atmosphere. No credit is taken for the containment as a pressure containing boundary and therefore unfiltered leak rate testing of the containment is no longer applicable. The dose at the site boundary associated with the events described in Attachment 1 of this letter does not exceed the requirements of 10 CFR 100.11 and the EPA PAGs.

Therefore, deletion of TS 3/4.1 "Containment" does not significantly reduce the margin of safety during Phase 1b and Phase 2.

TS 3/4.2 "Reactor Vessel Fuel" establishes a Safe Fuel Mass Limit (SFML) for the PDMS condition, which ensures that the amount of core debris that may be removed from the RV or rearranged in the RV during PDMS does not exceed 42kg. This SFML limit is specified to ensure subcriticality even after dual errors.

A calculation is presented as Attachment 5 (of Reference 1) which provides the basis to increase the SFML from 42 kg to 1200 kg.

The current SFML was developed based solely on credible upper bounds for input parameters as opposed to sample data or realistic conditions. The proposed revision to the SFML is based upon existing data and known conditions. These inputs are still considered to be reasonably and sufficiently conservative for their use in development of the proposed 1200 kg SFML. The derived SFML bounds the entire expected fissile mass inventory throughout all physically separated areas within the reactor building. The bounding fissile mass used to produce the SFML is assembled in idealized conditions that cannot credibly exist during decommissioning operations. Even if the expected remaining fissile mass throughout the building, including hold up in all piping and cubicles were to be brought together, a criticality is not feasible. There are no credible operational upsets to realize the ideal configuration but even in the event that the upset occurs, it would require fissile mass in excess of that analyzed, which is greater than what is anticipated. In addition, the SFML is based on a significantly reduced impurity concentration below that demonstrated to be present. The k_{eff} for the new SFML in the idealized static conditions does not exceed 0.95.

The calculation of the new SFML states that the entire mass of the core debris material cannot be configured into an arrangement whereby a criticality event is possible. Debris material removal operations will involve loading 12-14 storage casks with each cask containing less than the total SFML calculated for Phase 1b.

The overall subcritical nature, namely inherent elemental constituents, of the fuel debris remaining at the TMI-2 facility today is equivalent to that associated with the fuel debris at TMI-2 prior to defueling operations. The presence of some intact fuel, and the results of sampling campaigns conducted prior to defueling indicating slight impurity gradients through the RV did not easily allow the application of a representative fuel composition to the entirety of the core during the development of the previous SFML. Further, static and accident conditions analyzed after defueling merely credited the minimum concentration of impurities to ensure the facility was safe. In each of these scenarios, the applied conservatisms are different. Currently, core debris in the lower head region of the RV is most representative of what remains in the RV at the present time. Therefore, a reasonable representative impurity concentration can be applied to the homogenized mass in development of a new SFML for D&D. A conservative approach to adequately represent the inherent characteristics of the remaining fuel debris can be taken with respect to the

development of an SFML for the remaining decommissioning activities. This approach would not necessarily be applicable for the previous defueling operations or the related SFML developed at that time. The current SFML was conservatively derived and, coupled with the conservatively estimated masses and the planned decommissioning operations, provides significant and adequate margin of safety that ensures that the potential for a criticality is not credible.

The proposed change does not exceed or alter the SFML design basis as presented in the UFSAR and k_{eff} for the new SFML does not exceed 0.95. Therefore, the deletion of PDMS TS 3/4.2 "Reactor Vessel Fuel" does not involve a significant reduction in a margin of safety during Phase 1b and Phase 2.

As part of the PDMS condition, loads in excess of 50,000 lbs. are prohibited from travel over the RV.

The deletion of TS 3/4.3 does not exceed or alter a design basis or safety limit. As discussed in Section 2 "Detailed Description and Basis for The Changes," for Phase 1b and Phase 2, TMI-2 Solutions will develop a hoisting and rigging program that addresses movement of loads at TMI-2. The purpose of the hoisting and rigging program is to define the minimum requirements for the safe operations of cranes and hoists. The hoisting and rigging program will provide as applicable, detailed requirements for training and qualification of personnel, inspection and maintenance of cranes or hoists, the safe use of rigging equipment as well as direction for performing Non-Standard Lifts in order to ensure that lifting operations are performed in a safe manner. A lift plan will be developed for all lifts as directed by the hoisting and rigging program where a load drop or load impingement could contribute to release or dispersal of radioactive material to the environment could exceed the threshold for an unusual event.

Implementation of the hoisting and rigging program provides a defense in depth approach to preventing a load drop from occurring. Crane design features such as load cells, and travel stops, will be employed as required to ensure safe travel paths. Barriers will be provided as required to preclude the effects of a load drop.

A calculation has been performed, Attachment 5 (of Reference 1), that assesses increasing the Safe Fuel Mass Limit (SFML) from 42 kg to approximately 1200 kg. As stated in the calculation, it is not credible to have 1200 kg U in an idealized configuration for criticality to occur. There are no credible operational upsets to realize the ideal configuration but even in the event that the upset occurs, it would require fissile mass in excess of that analyzed, which is greater than what is anticipated, in addition to a greatly reduced impurity concentration to present a criticality hazard.

Therefore, the deletion of TS 3/4.3 "Crane Operations" does not significantly reduce the margin of safety during Phase 1b and Phase 2.

The TMI-2 sealed sources are maintained at TMI-1 and managed by Exelon under a

program compliant with the requirements of 10 CFR 70.39(c). Deleting TS 3/4.4 "Sealed Sources" from the TMI-2 TS and relocating the TS requirements to the DSAR does not involve a significant reduction in a margin of safety.

The proposed changes do not affect remaining plant operations, systems, or components supporting decommissioning activities. The proposed changes do not result in a change in initial conditions, or in any other parameter affecting the course of the remaining decommissioning activity accident analysis. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, TMI-2 Solutions concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

3.4 Conclusion

In conclusion, based on the considerations discussed above: 1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, 2) such activities will be conducted in compliance with the NRC's regulations, and 3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment 3
List of Regulatory Commitments

The table included in this attachment identifies the regulatory commitments in this document. The type of commitment and associated schedule for implementation are provided. Any other statements in this submittal represent intended or planned actions. They are provided for information purposes and are not considered to be regulatory commitments.

The following list identifies the action committed to by the Applicant identified in this letter. Any other actions discussed in the submittal represent intended or planned actions by the Applicant. They are described only as information and are not Regulatory Commitments.

REGULATORY COMMITMENT	TYPE		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
TMI-2 Solutions in coordination with Exelon will amend the TMI Emergency Plan to add Emergency Action Levels for Effluent Releases and Fire.	X		1/31/2022
TMI-2 will update the DQAP as described in this LAR	X		In conjunction with Implementation of this change
TMI-2 Solutions will develop a hoisting and rigging program that addresses movement of loads at TMI-2 and specify when lift plans are needed	X		Prior to entry into Phase 1B
The TMI-2 PDMS SAR will be updated to address the Events Analysis provided in	X		Prior to entry into Phase 1B

Attachment 3
List of Regulatory Commitments

Attachment 1 and be issued as a DSAR			
The Radiation Protection Plan (RPP) will identify the controls that will be implemented through procedures during D&D activities occurring inside of the RB.	X		Prior to entry into Phase 1B
Airborne radiation monitoring will be provided to monitor containment openings for airborne radioactivity.	X		Prior to entry into Phase 1B